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PSEG
Nuclear LLC

10CFR50.73

LR-N15-0240

NOV 24 2015

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Hope Creek Generating Station Unit 1
Renewed Facility Operating License No. NPF-57
Docket No. 50-354

Subject: Licensee Event Report 2015-005-00

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), PSEG Nuclear LLC is submitting the enclosed Licensee Event Report (LER) Number 2015-005-00, "Reactor Scram Due to Invalid RRCS Actuation."

If you have any questions or require additional information, please contact Mr. Thomas MacEwen at (856) 339-1097.

There are no regulatory commitments contained in this letter.

Sincerely,

Eric S. Carr
Plant Manager
Hope Creek Generating Station

ttm

Attachment: Licensee Event Report 2015-005-00 cc: Mr. Daniel Dorman, Regional Administrator – Region I, NRC

Mr. Tom Wengert, Project Manager - US NRC

Justin Hawkins, NRC Senior Resident Inspector – Hope Creek (X24)

Mr. Patrick Mulligan, Manager IV
Bureau of Nuclear Engineering
New Jersey Department of Environmental Protection
PO Box 420
Trenton, NJ 08625

Mr. Thomas MacEwen, Hope Creek Commitment Tracking Coordinator (H02)

Mr. Lee Marabella - Corporate Commitment Tracking Coordinator (N21)

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE Reactor Scram Due to Invalid RRCS Actuation

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	28	2015	2015	- 005	- 00	11	24	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMER *(Include Area Code)*
Thomas MacEwen, Lead Compliance Engineer	856-339-1097

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
■ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☐ NO		01	15	2016

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On September 28, 2015, at 20:46, with the Hope Creek reactor operating at 100% power, a human error during surveillance testing resulted in the actuation of the Redundant Reactivity Control System (RRCS), and subsequently, an automatic reactor scram on a valid low water level signal. At the time of the transient, a surveillance test of division 1 of the RRCS system was in progress. The test simulates a high reactor pressure signal. Plant data show the signal was entered in both channels of division 1 of the RRCS system. The resulting system actuation caused a trip of both Reactor Recirculation Pumps, and the actuation of the Alternate Rod Insertion (ARI) function of the RRCS system. As a result of these two actuations, reactor power lowered, causing reactor water level to lower to the Reactor Protection System (RPS) trip set point of +12.5 inches. The RPS initiated an automatic reactor scram. Reactor operators recovered water level to within the desired band using the feedwater system. Reactor pressure was maintained using turbine bypass valves discharging to the main condenser.

This report is being submitted under 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in the actuation of the Reactor Protection System. A cause evaluation is being conducted to determine the causes associated with the event. A supplement to this LER will be submitted to report the results of the cause evaluation.

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION
General Electric – Boiling Water Reactor (BWR/4)
Reactor Protection System – EIIS Identifier {JC}*
Redundant Reactivity Control System - EIIS Identifier {JC}*
Reactor Recirculation System - EIIS Identifier {AD}*

*Energy Industry Identification System {EIIS} codes and component function identifier codes appear as {SS/CCC}

IDENTIFICATION OF OCCURRENCE
Event Date: 09/28/15
Discovery Date: 09/28/15

CONDITIONS PRIOR TO OCCURRENCE
Hope Creek was in Operational Condition 1 at 100 percent rated thermal power (RTP). Redundant Reactivity Control System (RRCS) {JC}, Division 1, surveillance testing was in progress.

DESCRIPTION OF OCCURRRENCE
On 9/28/2015 at 20:46, a Hope Creek Instrument and Controls technician was performing a surveillance test of RRCS division 1, channel B, to simulate a high reactor pressure condition. The RRCS system is designed to detect and respond to an Anticipated Transient Without Scram (ATWS) condition. One indication of this condition is high reactor pressure, at or above 1071 psig. Under these conditions, the RRCS is designed to trip both Reactor Recirculation Pumps (RRPs) {AD} and initiate Alternate Rod Insertion (ARI). The RRPs are tripped to reduce core flow and increase the formation of core voids, thus reducing power. ARI provides an alternate path for control rod insertion by depressurizing the scram air header through valves separate from the RPS {JC} scram valves.

During the test, a keypad on the local RRCS panel is used to enter the test parameter, the test signal value and the channel being tested. The technician was expected to enter a test pressure signal of 1400 psig into the B channel of division 1. Plant data indicate the test pressure signal was also entered in channel A of division 1. With the 1400 psig test signal in both the A and B channels of logic, division 1 of the RRCS system actuated, causing RRPs to trip and ARI to begin control rod insertion by depressurizing the scram air header.

The change in reactor power caused a reactor water level transient which reached the RPS trip set-point of +12.5 inches. Although the control rods were already moving inward due to ARI actuation, the RPS functioned as designed to ensure reactor shutdown was completed via a scram signal. After the initial transient, plant operators stabilized reactor pressure and water level using turbine bypass valves and the feed water system, respectively.

CAUSE OF EVENT
A cause evaluation is being conducted to determine the causes associated with the event. A supplement to this LER will be submitted to report the results of the cause evaluation.

SAFETY CONSEQUENCES AND IMPLICATIONS
There were no consequences to nuclear safety as a result of this event. The RRCS and RPS system operated as designed to shut down the reactor. All necessary support systems functioned as needed to support plant stabilization and recovery post transient.

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NARRATIVE

SAFETY SYSTEM FUNCTIONAL FAILURE

A review of this condition determined that a Safety System Functional Failure (SSFF) as defined in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," did not occur.

PREVIOUS EVENTS

The cause evaluation will review similarity to previous events. The result of that review will be included in the supplement.

CORRECTIVE ACTIONS

The technician involved in the event was disqualified from performing any surveillance testing or other plant maintenance duties.

Other corrective actions will be determined by the cause evaluation.

COMMITMENTS

This LER contains no regulatory commitments.